DESIGN ADEQUACY TO MATERIAL

- STRENGTH
- TOUGHNESS (DUCTILITY)
- PRODUCEABILITY
- TESTABILITY
- CORROSION RESISTANCE
- REPAIRABILITY

Manufacturing Aspects

- Utilization of material simple in the application
- Product and wall thickness shall allow NDE
- The number of welds should be reduced

FURNAS ACTIVITIES IN SAFETY AND QUALITY ASSURANCE

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1. INTRODUCTION

Although FURNAS subcontracts all the required Architect Engineering tasks, it is a main company policy to have a strong participation during the development of the design and the construction of a Project. This participation is realized through technical review and discussion of all major activities of the Project.

Today we can say that FURNAS participation in the Nuclear Project is similar to the participation of the average USA Utilities, and greater than the majority of the German Utilities.

The reason for having adopted this philosophy was the need to prepare our technicians to handle problems associated with Nuclear Projects, problems that were largely unknown due to the pioneer nature of the Nuclear Projects in Brazil.

During the licensing of the plant our technical group had the opportunity of solving directly some of the problems as well as facilitate the interpretation of questions and answers made by the licensing authority and suppliers.

The recent incidents and new regulatory positions in the U.S.A., and other countries proved the soundness of the FURNAS philosophy of having a group with the knowledge and capability of quickly solving the problems not only related to design and construction but also those which will appear during the operational phase.

2. FURNAS' ACTIVITIES IN QA

Since the beginning of the ANGRA Nuclear Project, FURNAS was involved on activities related with QA. This was a new subject considering that ANGRA is the first Nuclear Power Plant in Brasil and through our consultants the QA requirements were included as contractural obligations of our main contractors

In 1972 FURNAS studied the QA philosophy in order to establish an organization inside the Company to implement the QA requirements in the design, fabrication and construction of the Nuclear Plant.

A QA organization to implement the QA requirements in the Project in accordance with the FURNAS philosophy was established in 1973.

To facilitate understanding the aspects involved, this presentation will be subdivided in four parts:

a) Licensing requirements
b) FURNAS' overall quality assurance program approach
c) Main practical aspects of the quality assurance requirements
d) Quantitative figures on FURNAS' overall quality assurance program.

2.1 Licensing requirements

The Brazilian licensing authority CNEN (Comissão Nacional de Energia Nuclear) adopted in 1977, for use in Brazilian nuclear installations, the IAEA Safety Code of Practice on Quality Assurance as the governing regulation for quality assurance activities.

For the Angra I Project, which was started prior to the adoption of the IAEA Criteria, the quality assurance program was designed to meet the intent of US NRC 10 CFR 50 Appendix B.

It should be noticed that the use of the two mentioned quality assurance criteria does not interfere with this presentation as their requirements are entirely consistent.

2.2 FURNAS' overall quality assurance approach

In accordance with the referred licensing quality assurance requirements, FURNAS; as the owner and operator of the plants, has the ultimate responsibility for the implementation of the overall quality assurance program.

Consistently with those requirements, FURNAS elect to delegate to its contractors the responsibility for the implementation of the quality assurance program parts that apply to their respective scopes of work.

In light of the above delegation, FURNAS' quality assurance activities are mainly directed to the monitoring of contractor's quality assurance program implementation and effectiveness. This is accomplished by means of periodic and sistematic audits and inspections which may be conducted either directly by FURNAS or by specialized consulting organizations.

2.3 Main practical aspects of the quality assurance requirements

The quality assurance requirements stipulated in the previously mentioned IAEA and US NRC governing documents provide the basic philosophy to be applied to all project activities, since design inception up to the end of construction and testing, in order to assure that safety related structures, components and systems will perform satisfactorily in service.

It should be emphasized that those requirements are composed of a condensed selection of good engineering practices that, when applied together, permit the achievement of the goal of the quality assurance program, which is not only to achieve the specified levels of quality but also to provide documentary evidence of levels achieved.

The referred governing documents include guidance in the following areas: structure of the QA programs, organization, document control, design control, procurement control, material control, process control, inspection and test control, nonconformance control, corrective actions, records control and audits.

Annex 1 summarizes the main aspects to be verified by FURNAS' auditors to assure compliance of contractors' QA programs with the guidelines dictated by the governing documents for each of the previously mentioned areas.

2.4 Quantitative figures of FURNAS' overall quality assurance program:

2.4.1 In order to ilustrate the involvement of FURNAS' personnel in the implementation of the overall quality assurance program, the following figures can be used. Figures related to site QA activities have been selected because they are simpler to quantify.

Site QA audits conducted

Angra 1 - from 1973 through 1979: 112
Angra 2 - from 1977 through 1980: 6

Site QA documented inspections

Angra 1 - from 1973 through 1980: 1088
Angra 2 - from 1978 through 1980: 51

2.4.2 In order to illustrate the involvement of FURNAS' constractors in the overall quality assurance program, the following figures can be used.

Erection Contractor (Angra 1)

Has implemented 18written QA procedures, 30 written QC procedures and 29 written QC work instruction (Annex 2 item 1).
Civil Contractor (Angra 1)

Civil Contractor (Angra 2)

Has implemented up to now 14 written QA procedures and 4 written QC procedures (Annex 2 item 3).

3. FURNAS ACTIVITIES IN SAFETY

3.1 Siting

FURNAS, as the responsible for the construction and operation of the three first nuclear power plants in Brazil, performed preliminary studies for site selection, as defined in CNEN Resolution 9/69, which is very similar to the American 10 CFR 100.

The Ministery of Mines and Energy authorized FURNAS to install the first Brazilian nuclear plant at Itaorna near Angra dos Reis in July 1970. Then, FURNAS began the following complementary investigations; sea water temperature measurements, detailed topographical surveys, determination of the land area, geological and seismologic studies, oceanographic studies (bathymetry, tides, currents, biota, etc), micro-meteorological and radiological studies. For this last study, FURNAS constructed a special laboratory near the site to perform all the pre-operational radiological studies.

The main purposes of these studies are: to furnish data to the architect-engineer in order to have a more safe design of the plant and to establish operational limites for radioactive discharges to the environment consistent with safeguarding the neighboring population.

At the same time these studies are used in order to identify critical human groups, critical food chains and to characterize the types of usage being made of the land and the nearby bodies of water. The survey also has the objective of collecting information pertinent to the planning of administrative controls which will be necessary to exercise over the area to assure the safety of the population in the unlikely event any radioactive releases occur which could result in exposure of the population outside the boundaries. In this case, FURNAS has already prepared an Emergency Plan which delineates all the actions that must be taken by FURNAS and State and Federal Authorities in an emergency.

At the beginning of 1974 the possibility of installing two more units at Itaorna became evident. With the decision by the Brazilian Government in favor of Angra Units 2 and 3, FURNAS initiated a program in order to supplement the site information and to review some site related items which could influence the design of the new units. Meteorological studies, for example, were re-evaluated it was decided to implement a new program.

It is important to point out that all the studies performed, or being performed by FURNAS, meet the most recent US NRC requirements and the most sophisticated US and Germany Plant pre-operational programs.

3.2 Licensing Safety Review

We have reviewed all licensing documents including Preliminary Safety Analysis Report and Final Safety Analysis Report. Based on our review these documents were modified prior to the submission to the Comissão Nacional de Energia Nuclear.

Besides these licensing documents, the overall mechanical, electrical and control design related to the plant safety was reviewed in order to check compliance with the applicable rules, criteria and regulatory guides.

For instance we can mention the review of the Réactor Pressure Vessel Design which we performed using the ASME Boiling and Pressure Vessel Code Section III, and all applicable code cases for class A Vessels and ASME Section IX welding qualification, and so on.

Regarding the nuclear design we can mention our review of the Angra I - Nuclear Design Report , which is of particular importance, because to check the results presented we performed calculations using our own methodology and discovered several discrepancies corrected by the designer.

Related to the accident and transient analysis we did a complete review of the documents issued by Westinghouse covering the following events, as required by Regulatory Guide 1.70:

1. Steady state and shutdown operations

   a. Power operation (- to 100 percent of full power)
   b. Startup (or standby) (critical, 0 to 15 percent of full power)
   c. Hot shutdown (subcritical, Residual Heat Removal System in isolation
   d. Cold shutdown (subcritical, Residual Heat Removal System in operation)
   e. Refueling.

2. Operation with permissible deviations

   a. Operation with components or systems out of service.
   b. Leakage from fuel with clad defects.
   c. Radioactivity in the reactor coolant:
      1) Fission products
      2) Corrosion products
      3) Tritium
   d. Operation with steam generator leaks up to the maximum allowed by Technical Specifications.
   e. Testing as allowed by the Technical Specifications.

3. Operational transientes

   a. Plant heatup and cooldown(up to 100^{o}F/hour for the Reactor Coolant System; 200^{o}F/hour for the pressurizer).
   b. Step load changes (up to $\pm$ 10 percent).
   c. Ramp load changes (up to 5 percent/minute).
   d. Load rejection up to and including design load rejection transient.

4. Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical Condition.

5. Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power.

6. Rod Cluster Control Assembly Misalignment.

7. Uncontrolled Boron Dilution.

8. Partial Loss of Forced Reactor Coolant Flow.

9. Start-up of an Inactive Reactor Coolant Loop.

10. Loss of External Electrical Load and/or Turbine Trip.

11. Loss of Normal Feedwater.

12. Loss of Off-Site Power to the Plant Auxiliaries (Plant Blackout).

13. Excessive Heat Removal Due to Feedwater System Mal-Functions.

14. Excessive Load Increase.

15. Accidental Depressurization of the Reactor Coolant System.

16. Accidental Depressurization of the Main Steam System.

17. Inadvertent Operation of Emergency Core Cooling System during Power Operation.

18. Loss of Reactor Coolant from Small Ruptured Pipes or from Cracks in Large Pipe which Actuate Emergency Core Cooling.

19. Inadvertent Loading of a Fuel Assembly into an improper Position.

20. Complete Loss of Forced Reactor Coolant Flow.

21. Waste Gas Decay Tank Rupture.

22. Single Rod Cluster Control Assembly Withdrawal at Full Power.

23. Major Rupture of Pipes containing reactor coolant up to and including double-ended rupture of the largest pope in the Reactor Coolant System (Loss of Coolant Accident).

24. Major Secondary System Pipe Rupture up to and including double-ended rupture (Rupture of a Steam Pipe).

25. Steam Generator Tube Rupture

26. Single Reactor Coolant Pump Locked Rotor.

27. Fuel Handling Accident.

28. Rupture of a Control Rod Mechanism Housing (RCCA Ejection)

3.3 Safety Analysis

During the review of the safety analysis documentation we have realized the necessity of, at least in some cases, to go deeper into this review, reproducing the results presented in the documents by means of our own calculations.

To perform this check of results it was necessary to define and implant a package of codes capable of simulating the plant behaviour during abnormal conditions. We have selected the RELAP-4 code, mainly for loss of coolant accidents, CONTEMPT-LT, for the Containment Pressure Analysis, and TOODEE-2 for the prediction of fuel behaviour during these accidents.

These codes were developed by NRC and presently are used for safety analysis by NRC in the USA, for auditing reactor vendor models, as well as by our Comissão Nacional de Energia Nuclear.

The RETRAN code, developed for EPRI, and used by several utilities in the USA, was selected for the simulation of Transients, small breaks and Steam Line Ruptures.

3.3.1 Codes Description

A brief description of these codes are presented below:

a) RELAP-4 is a computer program written in FORTRAN IV for the digital computer analysis of nuclear reactors and related systems. It is primarly applied in the study of system transient response to postulated perturbations such as coolant loop rupture, circulation pump failure, power excursions, etc. The program was written to be used for water-cooled (PWR and BWR) reactors and can be used for scale models such as LOFT and SEMISCALE. Additional versatility extends its usefulness to related applications, such as ice condenser and containment subcompartment analysis. Specific options are available for reflood (FLOOD) analysis and for the NRC Evaluation Model.

RELAP-4 models system fluid conditions including flow, pressure , mass inventory, fluid quality, and heat transfer. A subroutine provides water property tables. Component thermal conditions and energy transfers are modeled. The reactor system is subdivided into discrete volumes which, with interconnecting junctions (flow paths), are treated as one dimensional homogeneous elements. RELAP-4 solves an integral form of fluid conservation and state equations for each user defined volume and generates a time history of system conditions. Data are recorded for volume fluid, component heat, and junction flow characteristics. The output is in the form of printed tabular digital data. Available subroutines also allow output to be plotted as a function of time. Provision is made for selectively stopping the program at any point for data edits.

The program can be restarted for problem continuation.

b) CONTEMPT-LT is a digital computer program, written in FORTRAN IV, developed to describe the long-term behaviour of water-cooled nu- nuclear reactor containment systems subjected to postulated loss-of-coolant accident (LOCA) conditions. The program calculates the time variation of compartment pressures, temperatures, mass and energy inventories, heat structure temperature distributions, and energy exchange with adjacent compartments. The program is capa - ble of describing the effects of leakage on containment response. Models are provided to describe fan cooler and cooling spray en - gineered safety systems. Up to four compartments can be modeled with CONTEMPT-LT, and any compartment except the reactor system may have both a liquid pool region and an air-vapor atmosphere re gion above the pool. Each region is assumed to have a uniform tem perature , but the temperatures of the two regions may be diffe - rent. CONTEMPT-LT can be used to model all current boiling water reactor pressure system, including containments with either ver - tical or horizontal vent system. CONTEMPT-LT can also be used to model pressurized water reactor dry containments , subatmospheri containments, and dual volume containments with an annulus region, and can be used to describe containment responses in experimental containment system.

c) TOODEE-2 is a two dimensional, time dependent fuel element ther - mal analysis program developed by staff of the Nuclear Regulatory Commission from the TOODEE Code. Written in FORTRAN IV language , TOODEE-2 was developed primarily as an evaluation tool to calcu- late fuel element thermal response during the refill and reflood phase of a loss-of-coolant accident in a pressurized water reac - tor. For small breaks the code may be used for the entire tran - sient.

d) RETRAN represents a new computer code approach for analyzing the thermal-hydraulic response of Nuclear Steam Supply Systems (NSSS) to hypothetical Loss of Coolant Accidents (LOCA) and Operational Transients. In contrast to the "conservative" approach, RETRAN pro vides "best estimate" solutions to hypothetical LOCAs and Opera - tional Transients. RETRAN is a computer code package developed from the RELAP series of codes, from reference data, and from extensive analytical and experimental work previously conducted relative to the thermal-hydraulic behaviour of light-water reactor systems sub jected to postulated accidents and operational transient conditions. The RETRAN computer code is constructed in a semimodular and dynamic dimensioned form where additions to the code can be easily carried out as new and improved models are developed.

3.3.2 Work Development

Based on our own analysis we have issued the following reports:

- Cold Leg Loss of Coolant Accident during Blowdown and Refill Phases with Discharge Coefficient = 1.0.

- Cold Leg Loss of Coolant Accident during Blowdown and Refill Phases with disch Coefficient = 0.4.

- Cold Leg Loss of Coolant Accident during Reflood Phase with Discharge Coefficient = 0.4 and 1.0.

- Fuel Element Thermal Analysis for Cold Leg Loss of Coolant Accident during Refill and Reflood Phases with Discharge Coefficient = 0.4.

- Containment Pressure and Temperature Analysis for Suction Pump Line Break with Discharge Coefficient = 0.6.

Presently we are writing reports for the "Single Reactor Coolant Pump Locked Rotor", "Loss of Load and Turbine Trip" and "Uncontrolled Rod Cluster Control Assembly Bank Withdrawal" analyses. Besides we are working in models for "Samll Breaks" and "Steam Line Rupture" accidents and performing sensitivity studies for LOCA. The "Loss of Flow" "Station Blackout" transients will follow.

4. CONCLUSIONS

Based on our experience and comparing it with the trends in the utilities around the world, forced by economics reasons and/or licensing requirements, we can say that the Company is more and more directly involved with the problems of Quality and Safety.